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Abstract

THE THERMAL FLUX DISTRIBUTION IN THE MODERATOR REGION OF A SQUARE LATTICE OF CYLINDRICAL FUEL ELEMENTS

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The problem of deriving the mono-energetic flux distribution in a square lattice cell with a constant neutron source in the moderator region has been treated by several authors. Earlier studies neglected the absorption in the moderator, but more recently this has been taken into account. However, all these authors have only solved the problem in the diffusion approximation. Although it may be argued that the diffusion theory should hold in the moderator, this is certainly not true near the fuel rod. The exact flux shape near the fuel is important since it is mainly in this region that all extra absorbers, such as cans and coolant tubes, are situated. Also, one would like to know the true theoretical flux distribution, in order to be able to check and analyze flux measurements.

Correct flux distributions should, in principle, be calculated by one of the more or less sophisticated transport approximations. The purpose of the present investigation is to study a much simpler approach: one needs, essentially, only to solve the diffusion equation again; using the fuel disadvantage factor, F, and the extrapolation length at its surface, λ, this solution is easily corrected for transport effects.

The final solution is of the form

$$\Phi(x,y) = A[1 + \alpha_1 f(x,y; K_1) + \alpha_2 f(x,y; K_2)]$$

where f is the solution of the diffusion equation with the appro-

priate boundary conditions, K_1^2 is the reciprocal diffusion area, and K_2^2 is the second solution of the characteristic P_3 equation. The first two terms form the asymptotic (or diffusion) solution, and the third is the transport correction.

The coefficient α_1 is determined from λ, which is the reciprocal logarithmic derivative of the asymptotic flux. α_2 is determined by F, requiring flux continuity on the fuel-moderator boundary and the equality of the total absorption and source in the cell.

An ALTAC code has been written for a PHILCO-2000 computer, which gives the equi-flux curves in the moderator. These have been compared with a measured flux distribution in a natural uranium-heavy water lattice; the agreement is quite satisfactory.